ACCESSION #: 9908030156

NON-PUBLIC?: N

LICENSEE EVENT REPORT (LER)

FACILITY NAME: Edwin I. Hatch Nuclear Plant - Unit 2 PAGE: 1 OF 7

DOCKET NUMBER: 05000366

TITLE: Personnel Error and Inadequate Corrective Action Cause

Automatic Reactor Shutdown

EVENT DATE: 06/28/1999 LER #: 1999-007-00 REPORT DATE: 07/27/1999

OTHER FACILITIES INVOLVED: DOCKET NO: 05000

OPERATING MODE: 1 POWER LEVEL: 98

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR SECTION:

50.73(a)(2)(iv)

LICENSEE CONTACT FOR THIS LER:

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Nuclear Safety and Compliance Manager, Hatch

COMPONENT FAILURE DESCRIPTION:

CAUSE: SYSTEM: COMPONENT: MANUFACTURER:

REPORTABLE NPRDS:

SUPPLEMENTAL REPORT EXPECTED: No

ABSTRACT:

On 06/28/1999 at 0300 EDT, Unit 2 was in the Run mode at a power level of approximately 2708 CMWT (98 percent rated thermal power). Personnel were in the process of filling and venting the "B" cold reference leg keepfill system. At that time, the reactor shut down automatically and inboard Group 2 Primary Containment Isolation Valves (PCIVs) closed on low water level. Level decreased when Reactor Feedwater

Pump (RFP) flow rate decreased in response to a false high level signal. Following the shutdown, level continued to decrease due to void collapse from the rapid reduction in power resulting in closure of the Group 5 PCIVs. Level reached a minimum of approximately 45 inches below instrument zero causing automatic initiation of the Reactor Core Isolation Cooling and High Pressure Coolant Injection systems. The low level initiation signal cleared before either system could inject makeup water to the vessel. Secondary containment automatically isolated and all four trains of the Unit 1 and Unit 2 Standby Gas Treatment systems automatically started on low level. The RFPs restored level to normal within one minute. Pressure was at its normal value at the time of the reactor shutdown and decreased thereafter; no Safety/Relief Valves lifted nor were any required to lift.

This event was caused by personnel error and inadequate corrective action. Involved personnel were disciplined per the Positive Discipline Program and the calibration of a level instrument has been changed.

PLANT AND SYSTEM IDENTIFICATION

General Electric - Boiling Water Reactor

Energy Industry Identification System codes appear in the text as (EIIS

Code XX).

DESCRIPTION OF EVENT

On 06/28/1999 at 0300 EDT, Unit 2 was in the Run mode at a power level of approximately 2708 CMWT (98 percent rated thermal power). At that time, personnel were in the process of filling and venting the "B" cold reference leg keepfill system per subsection 7.4 of procedure 34GO-OPS-004-2S, "Nuclear Boiler Lineup and Reference Leg Backfill." The system had been partially isolated and drained on 06/26/1999 under Clearance 2-99-244 in order to replace needle valve 2B21-F538B. Valve 2B21-F538B was replaced on 06/26/1999 per Maintenance Work Order 2-99-2353 after personnel discovered on 06/23/1999 that the valve would not control keepfill system flow rate.

Personnel were filling and venting the "B" cold reference leg keepfill system so the system could be returned to service following the completion of corrective maintenance.

At 0300 EDT, the reactor shut down automatically on low reactor vessel water level. Water level decreased when the flow rate of the Reactor Feedwater Pumps (RFPs, EIIS Code SJ) decreased in response to a false high water level signal from water level instrument 2C32-N004B. The reference leg for this level instrument is connected to the "B" cold reference leg keepfill system. A false high water level signal from instrument 2C32-N004B indicated to master feedwater level controller 2C32-R600 that reactor vessel water level was too high. The controller therefore reduced RFP flow rate in an attempt to return sensed water level to its nominal setpoint of 37 inches above instrument zero. The reduction in RFP flow rate caused actual vessel water level to decrease. Personnel took manual control of the RFPs in an attempt to increase reactor vessel water level; however, they were unable to prevent water level from decreasing to the automatic shutdown setpoint of three inches above instrument zero.

The decrease in water level resulted in receipt of a Group 2 Primary Containment Isolation System (EIIS Code JM) isolation signal and closure of the inboard Group 2 Primary Containment Isolation Valves (EIIS Code JM) per design. Because of the false high water level sensed by the reactor vessel water level instruments providing trip signals to the outboard Group 2 Primary Containment Isolation Valves, only the inboard Group 2 valves received a low water level isolation signal. Following the automatic reactor shutdown, water level continued to decrease due to void collapse from the rapid reduction in power. The decrease in water level resulted in receipt of a Group 5 Primary Containment Isolation System isolation signal and closure of the Group 5 Primary Containment Isolation Valves per design. The inboard and outboard Primary Containment Isolation System functioned properly given the water levels sensed by the instruments that provide inputs to the respective isolation logic trip subsystems.

Water level reached a minimum of approximately 45 inches below instrument zero (113.44 inches above the top of the active fuel) resulting in automatic initiation of the Reactor Core Isolation Cooling (RCIC,

EIIS Code BN) and High Pressure Coolant Injection (HPCI, EIIS Code BJ) systems per design. However, the RFPs rapidly recovered reactor vessel water level- therefore, the low water level initiation signal cleared before either system could inject makeup water to the reactor vessel.

Secondary containment automatically isolated and all four trains of the Unit 1 and Unit 2 Standby Gas Treatment (EIIS Code BH) systems automatically started on low water level per design. The RFPs increased water level to above 37 inches above instrument zero within one minute of the automatic reactor shutdown. The RFPs and the HPCI and RCIC systems, operating on minimum flow, subsequently tripped on high reactor water level. The reduction in feedwater flow rate caused by the trip of the RFPs resulted in the runback of both reactor recirculation pumps (EIIS Code AD) to minimum speed. (Personnel manually reduced the speed of the "A" recirculation pump when they noted the "B" pump had reduced speed automatically and the "N" pump had not. Subsequent investigation found the runback logic time delay relay settings were such that the "B" recirculation pump received the runback signal approximately 15 seconds sooner than the "A" pump. Personnel manually reduced "A" pump speed before its time delay relay could generate the automatic runback signal.)

Reactor vessel pressure was within its normal range at approximately 1030 psig at the time of the reactor shutdown. Pressure decreased immediately following the automatic reactor shutdown. The Main Turbine Bypass Valves (EIIS Code SO) automatically controlled reactor pressure. No Safety/Relief Valves lifted nor were any required to lift to reduce or control pressure.

CAUSE OF EVENT

This event was caused by personnel error and inadequate corrective action. Personnel filling and venting the keepfill system skipped a procedure step. This error led to some instruments sensing a false high water level. A long-standing instrument error in the reading of a water level instrument used in the median level signal processor resulted in the input of a false high water level to the master feedwater level controller. These problems caused the feedwater level control system to decrease the flow rate of the RFPs resulting in a decrease in actual vessel water level and an automatic reactor shutdown on low water level.

Personnel filling and venting the "B" cold reference leg keepfill system on 06/28/1999 skipped a step in subsection 7.4 of procedure 34GO-OPS-004-2S. This step required the keepfill system to be isolated from its associated reference leg. As a result of this error, the system was not isolated from its reference leg and personnel inadvertently vented the reference leg when they opened one of the keepfill system vent valves as part of filling and venting activities. Venting the reference leg caused the differential pressure across the level transmitters connected to that reference leg to decrease. The transmitters sensed the decreasing differential pressure as increasing reactor water level; therefore, the water level instruments on the affected reference leg indicated a false increasing water level.

One of the water level instruments connected to the vented reference leg inputs to median level signal processor 2C32-K648; this processor is part of the feedwater level control system. The processor's

purpose is to select which of three water level instrument signals will be sent to the master feedwater level controller. The master controller uses this level signal, along with feedwater and steam flow rate signals to control reactor water level at the nominal setting of 37 inches above instrument zero. The median level signal processor goes through a series of logic steps ten times every second in an attempt to determine and select the most accurate of the three level signals. A flowchart of the processor logic steps is given in Figure 1:

FIGURE 1 "2C32-K648 OUTPUT FLOWCHART" omitted

When the reference leg was vented, the reading from the "B" water level instrument began to increase. Its sensed water level deviated by more than five inches from the other two water level instruments, one of which supplied the median signal due to the problem with the "B" instrument. (The other two instruments are connected to an unaffected reference leg and continued to sense actual water level.) Therefore, the output of each of the first two logic steps shown in Figure 1 was negative. Because the venting of the reference leg was gradual, the "B" water level instrument did not fail upscale immediately. Therefore, the output of the third logic step of Figure 1 also was negative. Finally, because of a long-standing error in the reading of comparison water level instrument 2C32-R655 (its reading typically is not within ten inches of the "X" and "C" water level instruments), the output of the last logic step of Figure 1 was negative. Consequently, the output of median level signal processor 2C32-K648 was the signal from the "B" water

level instrument and a false high water level signal was input to the master feedwater level controller. This caused the master controller to decrease the flow rate of the RFPs in an attempt to reduce the false high water level condition. The automatic controller action resulted in a decrease in actual vessel water level and a subsequent automatic reactor shutdown on low water level.

REPORTABILITY ANALYSIS AND SAFETY ASSESSMENT

This report is required by 10 CFR 50.73 (a)(2)(iv) because of the unplanned actuation of Engineered Safety Feature systems. The Reactor Protection System (EIIS Code JC) and the Group 2 Primary Containment Isolation System, Engineered Safety Feature systems, actuated on low reactor water level. Following the automatic reactor shutdown, water level decreased due to void collapse. The decrease in water level resulted in the receipt of an automatic Group 5 Primary Containment Isolation System isolation signal on low reactor water level. The low reactor water level also caused automatic initiations of the RCIC and HPCI systems. The low level initiation signal cleared, however, before either system could inject makeup water to the vessel. Secondary containment automatically isolated and all four trains of the Unit 1 and Unit 2 Standby Gas Treatment systems automatically started on low level.

Low reactor vessel water level, initiated from four level transmitters that sense the difference between the pressures due to a constant column of water (reference leg) and the actual water level (variable leg) in the vessel, indicates the capability to cool the fuel may be threatened. Should water level decrease too far, fuel damage could result. Therefore, an automatic reactor shutdown is initiated to reduce substantially the heat generated in the fuel from fission. The automatic reactor shutdown reduces the amount of energy required to be absorbed and, along with the actions of the emergency core cooling systems, ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46.

The HPCI system is an emergency core cooling system designed to operate in conjunction with the reactor protection system. It is provided to ensure that the reactor is adequately cooled to limit fuel-clad temperature in the event of a small break in the nuclear system and a loss of coolant that does not result in rapid depressurization of the reactor vessel. The HPCI system permits the plant to be shut down while maintaining sufficient reactor vessel water inventory until the reactor vessel is depressurized.

The HPCI system continues to operate until reactor vessel pressure is below the pressure at which other emergency core cooling systems can maintain cooling.

To provide timely protection against the onset and consequences of accidents involving the gross release of radioactive materials from the fuel and nuclear system process barriers, the Primary Containment Isolation System initiates automatic isolation of lines which penetrate the primary containment whenever monitored variables exceed operational limits. A low water level in the reactor vessel could indicate that either coolant is being lost through a breach in the nuclear system process barrier or the normal supply of reactor feedwater has been lost and that the core is in danger of becoming overheated. Low reactor vessel water level initiates closure of various primary containment isolation valves. The closure of these valves is intended to isolate a line breach, conserve reactor coolant, and prevent the escape of radioactive materials from the primary containment.

In this event, the reactor shut down automatically and inboard Group 2 Primary Containment Isolation Valves closed on low reactor vessel water level. The low water level condition resulted from a decrease in feedwater flow rate caused by a false high water level input to the feedwater level control logic system. Group 5 Primary Containment Isolation Valves closed, the HPCI and RCIC systems initiated, the secondary containment isolated, and the Standby Gas Treatment system trains started at their respective setpoints as water level continued to decrease due to void collapse from the rapid decrease in reactor power.

All systems functioned as expected and per their design given the water level transient. The low water level condition was not the result of a line breach or escape of reactor coolant and there was no release of radioactive materials into either the primary or secondary containments. Water level was maintained well above the top of the active fuel throughout the transient and was restored to normal within one minute of the automatic reactor shutdown. Therefore, it is concluded the event had no adverse impact on nuclear safety. This analysis is applicable to all power levels.

CORRECTIVE ACTIONS

Personnel involved in filling and venting the "B" cold reference leg keepfill system were disciplined per the Positive Discipline Program.

The calibration of comparison water level instrument 2C32-R655 has been changed to ensure its reading normally is within ten inches of the readings of the three water level instruments that provide an input to median level signal processor 2C32-K648.

The reading of Unit 1 comparison water level instrument 1C32-R655 typically is within ten inches of the readings of the three water level instruments that provide an input to median level signal processor 1C32-K648.

Therefore, a similar problem does not exist on Unit I and no corrective actions are necessary.

ADDITIONAL INFORMATION

No systems other than those already mentioned in this report were affected by this event.

This LER does not contain any permanent licensing commitments.

No failed component caused or resulted from this event.

Previous similar events in the last two years in which the reactor shutdown from power were reported in the following Licensee Event Reports:

50-321/1999-003, dated 06/01/1999,

50-366/1999-005, dated 05/27/1999, and

50-366/1999-006, dated 07/14/1999.

The previous similar events were caused by, respectively, personnel error while implementing a clearance; manufacturer error and a combination of factors, including failure to follow procedure, inadequate and untimely corrective action, and inadequate design. The causes of the first two events were different from the causes of this event; therefore, their corrective actions could not be expected to have prevented this event. Some of the causes of the third previous event are similar in nature to the causes of this event. However, corrective actions for the last event had not been implemented at the time of this event because the events occurred within two weeks of each other. Therefore, the corrective actions for the last event could not have prevented this event.

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